June 2, 2009

NG-09-0430
10 CFR 50.73

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Licensee Event Report #2009-003-00

Please find attached the subject report submitted in accordance with 10 CFR 50.73. This letter makes no new commitments or changes to any existing commitments.

Richard L. Anderson
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Resident Inspector, DAEC, USNRC

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NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 8/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE

Unplanned Manual Scram due to Increasing Reactor Water Level

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER 05000
04	03	09	2009	003	0	06	02	09	FACILITY NAME	DOCUMENT NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (*Check all that apply*)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100%	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	VOLUNTARY LER

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (*Include Area Code*)
Bob Murrell, Engineering Analyst	319-851-7900

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (*If yes, complete 15. EXPECTED SUBMISSION DATE*) ☒ NO				

ABSTRACT (*Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines*)

On April 3, 2009, while operating at 100% power, with no Technical Specification (TS) Required Actions in affect, instrument and controls (I&C) technicians were performing Surveillance Test Procedure (STP) 3.3.3.2-09, Reactor Water Level and Pressure Instruments Calibration. This STP was performed for the first time since the recorder had been replaced during Refueling Outage (RFO) 21 in February 2009. Step 7.1.20 directed lifting a lead on the positive terminal of channel 2 on Level Recorder Switch (LRS) 4559/4560. When the lead was lifted, the control loop for reactor vessel level control was opened, and indicated vessel level pegged low. The plant response, as designed, was to increase feed water flow to the reactor to compensate for the indicated lowering vessel level. This caused actual reactor vessel level to increase. As a result of the increasing reactor water level, operators inserted a manual scram at 0028. All safety systems responded as designed. Containment isolations signals were received for groups 2, 3, and 4 due to reactor level dropping below 170 inches. All isolations went to completion. This reactor level response is normal following a reactor scram.

The cause of this event was an inadequate procedure change.

There were no actual safety consequences and no effect on public health and safety as a result of this event.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

I. Description of Event:

Generic Engineering Change Package (GMCP) 1751 was issued in August 2007 for the replacement of multiple analog paper recorders with digital paperless recorders. In support of this, Modification Work Order (MWO) 1143804 was written in January 2008 to replace LRS-4559/4560. The selected new recorders are Yokogawa DX100N series, and are designed for nuclear applications in that they can be removed from their assembly without interrupting a control loop.

On February 12, 2009, during RFO 21, the recorder was replaced and tested satisfactorily by I&C technicians. The construction acceptance testing for the modification involved checking the response of the recorder and associated alarms after it was installed in panel 1C-05 by lifting leads on the back of the panel.

On April 1, 2009, while reviewing STP 3.3.3.2-09, Reactor Water Level and Pressure Instruments Calibration, in preparations for upcoming testing, I&C technicians identified a step in the STP that needed clarification. Specifically, the I&C technician assigned to perform the STP realized that step 7.1.20, which directed the technician to remove the recorder from service, was vague since the step did not provide further guidance as to how the recorder removal is accomplished. This STP is performed on a 6 month frequency to fulfill the requirements of Technical Requirements Manual Surveillance Requirement 3.3.6.3. Therefore, this was the first performance of the STP since LRS-4559/4560 was replaced during RFO 21.

On April 2, 2009, after discussions with the procedures department and I&C supervision, the STP was revised to add more specific guidance to step 7.1.20. Step 7.1.20 was revised to read "Lift and tape the lead attached to the positive (+) terminal of channel 2 (black channel) on LRS-4559/60 (REACTOR WATER LEVEL recorder)." This was the same method that was used during RFO 21 to perform the construction acceptance testing. Neither the I&C technician, procedure writer or I&C supervision reviewed the system logic prints to make this revision; they solely relied on the acceptance testing for the MWO that made the modification to determine which lead to lift.

On April 3, 2009, with the plant operating at 100% power with no TS Required Actions in affect, I&C technicians, while performing STP 3.3.3.2-09, lifted the lead attached to the positive terminal of channel on LRS-4559/4560 as directed by step 7.1.20. When the lead was lifted, the control loop for reactor vessel level control was opened, and indicated vessel level pegged low. As designed, the plant response was to increase feed water flow to the reactor to compensate for the indicated lowering vessel level, which caused reactor vessel level to rise. As a result of the increasing reactor water level, a manual scram was inserted at 0028.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

II. Assessment of Safety Consequences:

No emergency core cooling system actuations occurred, or were required, as a result of this event. Additionally, the prompt operator action to insert the manual scram due to rising reactor water level, lowered reactor power sufficiently so that when the turbine tripped, steam flow was within bypass valve capacity, and safety relief valve actuations were not required to control reactor pressure.

There were no structures, systems, or components (SSCs) inoperable at the start of this event and no SSCs contributed to this event. Containment isolations signals were received for groups 2, 3, and 4 due to reactor level dropping below 170 inches. All isolations went to completion. This reactor level response is normal following a reactor scram.

This event did not result in a Safety System Functional Failure.

Therefore, the reactor scram did not result in any radiological or nuclear concern which would impact the health and safety of the public.

III. Cause of Event:

A root cause evaluation (RCE 1081) was completed for this event. The RCE determined that the revision to STP 3.3.3.2-09 introduced a latent error that removed the recorder from service, and interrupted the control loop. The specific root causes (RC) and contributing factors (CF) are as follows:

RC1- Electrical termination changes in STP's are not reviewed with the same requirements as maintenance activities.

RC2 – The site modification process does not require review of all service requirements, including how equipment is to be calibrated and tested while the unit is operating.

CF1- MWO 1143804 was used as the reference for which lead to lift, instead of the instrument drawings.

CF2- MWO 1143804 preconditioned the technicians to believe there was no adverse effect from de-terminating the recorder due to past acceptable performance during construction acceptance testing.

CF3- GMCP 1751 did not identify the different configuration for LRS-4559/4560.

CF4- Numerous STP's had been revised and the recorders calibrated using this removal methodology (Lifting Leads) with no adverse effects.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

IV. Corrective Actions:

Immediate Actions Taken

Engineered Maintenance Action (EMA) 92509 changed the configuration of LRS-4559/4560 to allow the recorder to be removed from service without interrupting the reactor vessel level control loop while the plant is operating at power.

Long Term Corrective Actions to Prevent Recurrence

Administrate Control Procedure (ACP) 106.1, Procedure Preparation, Revision, Review and Approval will be revised to require a documented plant effect evaluation whenever a step is revised that manipulates plant equipment

The site modification process will be revised to require a review of all service requirements of modified equipment, including operational requirements and online calibrations.

Other Corrective Actions

ACP 106.1 will be revised to add a new requirement to ACP 106.1, Section 3.14.2, Cross-Discipline Reviewer, stating that all procedure changes generated in support of design modifications require a cross discipline review by a design engineer prior to approval.

ACP 106.1 will be revised to require a Condition Report (CAP) be written anytime a procedure change is requested in less than 24 hours.

V. Additional Information:

Previous Similar Occurrences:

A review of LERs over the previous 5 years revealed the following similar occurrences:

LER 2007-007 - Reactor Scram Due to 1A2 Non-essential Bus Lockout
LER 2007-006 - Reactor Shutdown as a Result of a Chemistry Excursion
LER 2006-005 - Reactor Scram During Main Turbine Testing
LER 2009-001 - Manual Reactor Scram Due to Loss of Condenser Cooling

EIIS System and Component Codes:

JB - Feedwater/Steam Generator Water Level Control System

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Reporting Requirements:

This event is reportable under 10 CFR 50.73(a)(2)(iv)(A).